NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (EMEB)[1]

Secondary - None

I. AREAS OF REVIEW

Reactor pressure vessel internals consist of all the structural and mechanical elements inside the reactor vessel. General Design Criteria 1, 2, 4 and 10 and 10 CFR Part 50, §50.55a require that structures and components important to safety shall be constructed and tested to quality standards[2] commensurate with the importance of the safety functions to be performed, and designed with appropriate margins to withstand effects of anticipated operational occurrences, normal ~~plant operational occurrences~~operation;[3] natural phenomena such as earthquakes; postulated accidents including loss-of-coolant accidents (LOCA), and from events and conditions outside the nuclear power unit.

For the purpose of this standard review plan section, the term "reactor internals" includes core support structures and other internal structures and refers to all structural and mechanical elements inside the reactor pressure vessel with the exception of the following:

1. Reactor fuel elements~~,~~ and[4] the reactivity control elements out to the coupling interfaces with the drive units (the fuel system design is covered in Standard Review Plan (SRP) Section 4.2, but the structural aspects of reactor fuel assemblies are reviewed with the reactor internals).

2. Control rod drive elements (the drive elements inside the guide tubes are covered in SRP Section 3.9.4, but the guide tubes are reviewed with the reactor internals).

3. In-core instrumentation (in-core instrumentation support structures are reviewed with the reactor internals).

The staff review includes the following specific areas:

a. The physical or design arrangements of all reactor internals structures, components, assemblies, and systems ~~should be presented~~[5], including the manner of positioning and securing such items within the reactor pressure vessel, the manner of providing for axial and lateral retention and support of the internals assemblies and components, and the manner of accommodating dimensional changes due to thermal and other effects.

b. The loading conditions that provide the basis for the design of the reactor internals to sustain normal operation, anticipated operational occurrences, postulated accidents, and seismic events ~~should be specified~~[6]. All combinations of design and service loadings should be listed (e.g., operating pressure differences and thermal effects, seismic loads, and transient pressure loads associated with postulated loss-of-coolant accidents) that are accounted for in design of the reactor internals.

c. The design bases for the mechanical design of the reactor vessel internals, ~~should be presented~~[7] including allowable limits such as maximum allowable stresses; stability under dynamic loads; deflection, cycling, and fatigue limits; and core mechanical and thermal restraints (positioning and holddown). Details of dynamic analyses, input forcing functions, and response to loadings are discussed in SRP Section 3.9.2.

d. Each combination of design and service loadings, ~~should be~~[8] categorized with respect to the allowable design or service limits ~~(~~[defined in the ASME Code (Reference 7)[9] and SRP Section 3.9.~~5~~3[10]~~, Reference 5 and 7)~~][11], and the associated stress intensity or deformation limits ~~should be stipulated~~[12]. Design or service loadings should include safe shutdown earthquake (SSE) and operating basis earthquake (OBE)[13] loads as appropriate.

Review Interfaces

EMEB also performs the following reviews under the SRP sections indicated:[14]

1. Evaluates the rupture locations, rupture loads, and dynamic effects associated with the postulated rupture of piping as part of its primary review responsibility for SRP Section 3.6.2.[15]

2. Evaluates the adequacy of analysis methods for seismic Category I reactor pressure vessel internals and system dynamic analysis, identification of design transients, and identification of service lifetime transient cyclic loadings to be reflected in the design and fatigue analyses of reactor pressure vessel internals, as part of its primary review responsibility for SRP Section 3.9.1.[1617]

3. Evaluates the adequacy of dynamic analyses and proposed flow-induced vibration testing for reactor pressure vessel internals as part of its primary review responsibility for SRP Section 3.9.2.[18]

4. Evaluates the adequacy of the design for structural integrity of the reactor pressure vessel internals, including the adequacy of design fatigue curves for reactor internals materials with respect to cumulative reactor service-related environmental and usage factor effects and consideration of each combination of design, service, and postulated event loadings, as part of its primary review responsibility for SRP Section 3.9.3.[19][20]

5. Evaluates the adequacy of the mechanical design of the control rod drive system (CRDS), including the control rod drive elements, as part of its primary review responsibility for SRP Section 3.9.4.[21]

In addition, the EMEB[22] will coordinate other branches' evaluations that interface with the overall review of the reactor internals as follows:

1. The ~~Core Performance Branch (CPB)~~Reactor Systems Branch (SRXB)[23] will verify fuel system design, including fuel behavior effects ~~to~~on[24] reactor core design under various normal and accident operating conditions in SRP Section 4.2.

2. The Materials and Chemical Engineering Branch (~~MTEB~~EMCB)[25] will review material aspects of reactor internals in SRP Section 4.5.2. The EMCB evaluates the adequacy of analyses justifying exclusion of certain postulated pipe ruptures from design bases in SRP Section 3.6.3 (proposed).[26] The EMCB also reviews the adequacy of programs for assuring the integrity of bolting and threaded fasteners as part of its primary review responsibility for SRP Section 3.13 (proposed).[27]

For those areas of review identified above as ~~part of the primary review responsibility of other branches~~part of the review under other SRP sections, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP sections~~ of the corresponding primary branch~~.[28]

II. ACCEPTANCE CRITERIA

EMEB[29] acceptance criteria are based on meeting the requirements of the following regulations:

1. General Design Criterion 1 and 10 CFR Part 50, §50.55a, as ~~it~~they relate~~s~~ to reactor internals, require~~s~~[30] that the reactor internals shall be designed to quality standards commensurate with the importance of the safety functions to be performed.

2. General Design Criterion 2, as it relates to reactor internals, requires that the reactor internals shall be designed to withstand the effects of earthquakes without loss of capability to perform ~~its~~their[31] safety functions.

3. General Design Criterion 4, as it relates to reactor internals, requires that reactor internals shall be designed to accommodate the effects of and to be compatible with the

environmental conditions associated with normal operations, maintenance, testing, and postulated LOCA. Dynamic effects associated with postulated pipe ruptures may be excluded from the design basis when analyses demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.[32]

4. General Design Criteria 10, as it relates to reactor internals, requires that reactor internals shall be designed with ~~adequate margins to assure specified acceptable fuel design limits are not exceeded during anticipated normal operational occurrences.~~appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.[33]

Specific criteria necessary to meet the relevant requirements of the regulations identified above are as follows:

a. Requirements for loads, loading combinations, and limits applicable to those portions of reactor internals constructed to Subsection NG of the ASME Code are presented in SRP Section 3.9.3 ~~(Ref. 7)~~[34].

b. The design and construction of the core support structures should conform to the requirements of Subsection NG, "Core Support Structures," of the ASME Code ~~(Ref. 5)~~[35], and SRP Section 3.9.3.

c. The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals other than the core support structures should meet the guidelines of NG-3000 and be constructed so as not to adversely affect the integrity of the core support structures (NG-1122).

d. Deformation limits for reactor internals should be established by the applicant and presented in ~~his~~the[36] safety analysis report. The basis for these limits should be included. The stresses associated with these displacements should not exceed the specified limits. The requirements for dynamic analysis of these components are discussed in SRP Section 3.9.2.

e. The reactor internals should be designed to accommodate asymmetric blowdown loads due to postulated pipe ruptures. The applicant's evaluation of such loads should demonstrate that the loads do not exceed the limits imposed by the applicable codes and standards. Where double-ended guillotine break of reactor coolant piping is postulated, criteria for evaluating loading transients and structural components are specified in NUREG-0609 (Reference 6).[37]

Technical Rationale[38]

The technical rationale for application of the above acceptance criteria to the reactor pressure vessel internals is discussed in the following paragraphs:

1. GDC 1 and 10 CFR Part 50, §50.55a require that structures, systems, and components (SSCs) important to safety be designed to quality standards commensurate with the importance of the safety functions to be performed. The reactor internals include SSCs that perform safety functions and/or whose failure could affect the performance of safety functions by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the primary reactor coolant system). Application of this requirement to the reactor internals provides assurance that established standard design practices of proven or demonstrated effectiveness are used to achieve a high likelihood that these safety functions will be performed.

2. GDC 2, in relevant part, requires that SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The reactor internals perform and/or may affect the performance (through their failure) of safety functions including core cooling and fission product confinement. Application of GDC 2 to the reactor internals provides assurance that they will withstand earthquakes without damaging fuel cladding or interfering with core cooling.

3. GDC 4, in relevant part, requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, including LOCAs. The reactor internals perform and/or may affect the performance (through their failure) of safety functions including reactivity monitoring and control, core cooling, and fission product confinement. Application of GDC 4 to the reactor internals provides assurance that the effects of environmental conditions to which they are exposed over their installed life will not diminish the likelihood of performance of these safety functions under all operating conditions, including accidents. This provides assurance that failures of the reactor internals resulting from environmental service conditions that could cause loss of capability to monitor reactivity, fuel damage resulting from loss of reactivity control, structural damage to fuel cladding, or interference with core cooling are not likely to occur.

   NUREG-0609 identifies and evaluates certain postulated pipe ruptures (e.g. double-ended guillotine breaks of primary reactor coolant loop piping) that are known to result in asymmetric blowdown loadings on the reactor internals. GDC 4 allows such dynamic effects associated with postulated pipe ruptures to be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. Application of GDC 4 to the reactor internals provides assurance that asymmetric loading effects associated with postulated pipe ruptures are either accommodated in the design (with assurance of the functionality and integrity of reactor internals) or demonstrated to be extremely unlikely to occur. This provides assurance that overstress failures of the reactor internals that could cause loss of capability to monitor reactivity, fuel damage resulting from loss of reactivity control, structural damage to fuel cladding, or interference with core cooling are unlikely to occur.

4. GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The reactor internals perform and/or may affect the performance (through their failure) of safety functions including reactivity control and core cooling. These safety functions are essential to assure that specified acceptable fuel design limits are not exceeded. Application of GDC 10 to the reactor internals provides assurance that they are designed with sufficient margin to ensure their functionality and integrity during any condition of normal operation, including the effects of anticipated operational occurrences, such that a high likelihood of performance of these safety functions is achieved. Assured performance of these safety functions in turn assures that specified acceptable fuel design limits related to reactivity control and core cooling are not exceeded, thus assuring the integrity of the fuel and its cladding.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case.

The configuration and general arrangement of all mechanical and structural internal elements covered by this SRP section are reviewed and compared to those of previously licensed similar plants. Any significant changes in design are noted and the applicant is asked to verify that these changes do not affect the flow-induced vibration test results required by SRP Section 3.9.2.

With respect to the design and analysis of reactor internals, a statement by the applicant that they are designed in accordance with Subsection NG of the ASME Code and SRP Section 3.9.3, "Core Support Structures," ~~of Reference 5 and 7~~ is acceptable[39]. In lieu of such a commitment, the reviewer must determine that the design and analysis of these components are consistent with the requirements discussed in subsection II, above. This is accomplished by requiring that the applicant describe the design procedures and criteria used in the design of these components. This includes a list of the design and service stress limits used for all of the applicable loading conditions.

The reviewer verifies that the asymmetric blowdown loadings upon reactor internals resulting from pipe ruptures (at postulated locations not excluded based upon leak-before-break analyses) have been evaluated by the applicant and are accommodated in the design, consistent with criteria identified in specific criterion II.e.[40]

The deformation limits specified for these components are reviewed to verify that the applicant has stated that these deflections will not interfere with the functioning of related components, e.g., control rods and standby cooling systems, and that the stresses associated with these displacements are less than the specified limits for the core support structures.

At the operating license stage, the calculated stresses and deformations are reviewed to determine that they do not exceed the specified limits.

Any deviations that have not been adequately justified are identified and findings to that effect are transmitted to the applicant with a request for conformance with the requirements discussed in subsection II, above, or additional technical justification.

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[41]

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with this SRP section and that ~~his~~the[42] evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

> The staff concludes that the design of reactor internals is acceptable and meets the requirements of General Design Criteria 1, 2, 4, and 10 and 10 CFR Part 50, §50.55a. This conclusion is based on the following:
>
> 1. The applicant has met the requirements of GDC 1 and 10 CFR Part 50, §50.55a with respect to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the reactor internals are in conformance with the requirements of Subsection NG of the ASME Code, Section III.
>
> 2. The applicant has met the requirements of GDCs[43] 2, 4, and 10 with respect to designing components important to safety to withstand the effects of earthquakes[44] and the effects of normal operation, maintenance, testing, and postulated ~~loss-of-coolant~~ accidents (including LOCAs) with sufficient margin to assure that capability to perform ~~its~~their[45] safety functions is maintained ~~and the specified acceptable fuel design limits are not exceeded~~. The applicant has also designed the reactor internals with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.[46]
>
>   The specified design transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals structures and components provide reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components would not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand

the most adverse loading events which have been postulated to occur during service lifetime without loss of structural integrity or impairment of function.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP Section.[47]

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[48] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[49]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulations.[50]

VI. REFERENCES[51]

1. 10 CFR Part 50, §50.55a, "Codes and Standards."[52]

~~1~~2. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

~~2~~3. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bas~~i~~es[53] for Protection Against Natural Phenomena."

~~3~~4. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and ~~Missile~~Dynamic Effects Design Bases."[54]

~~4~~5. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

6. NUREG-0609; "Asymmetric Blowdown Loads on PWR Primary Systems: Resolution of Generic Task Action Plan A-2;" Hosford, S.B.; Mattu, R.; Meyer, R.O.; Division of Safety Technology; January, 1981.[55]

~~5~~7. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

~~6. Standard Review Plan Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment."~~

~~7. Standard Review Plan Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."~~[56]

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SRP Draft Section 3.9.5

Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB names and abbreviations	Editorial change made to reflect the current SRP Section 3.9.5 PRB abbreviation for the Mechanical Engineering Branch.
2.	Editorial	Added plural to improve grammar.
3.	Editorial	Revised to reflect the actual requirements of GDCs. The term "anticipated normal plant operational occurrences" is not consistent with terminology used in GDC 10 and is thus replaced with "anticipated operational occurrences." Also revised to include the term "normal operation" which is consistent with terminology used in GDC 4 and conveys similar meaning to the existing wording used in SRP Section 3.9.5, subsection I.
4.	Editorial	Grammar improvement-added conjunction since only two different items (fuel elements and reactivity control elements) are discussed.
5.	Editorial	Removed application content recommendations (apparently from Reg. Guide 1.70) to more clearly list specific areas of staff reviews.
6.	Editorial	Removed application content recommendations (apparently from Reg. Guide 1.70) to more clearly list specific areas of staff reviews.
7.	Editorial	Removed application content recommendations (apparently from Reg. Guide 1.70) to more clearly list specific areas of staff reviews.
8.	Editorial	Removed application content recommendations (apparently from Reg. Guide 1.70) to more clearly list specific areas of staff reviews.
9.	SRP-UDP format item.	Format change to make the the initial citation of references consistent with the SRP-UDP format guidance.
10.	Editorial	Editorial change made to clarify the existing intended citation of Reference 7 (i.e. SRP Section 3.9.3). Note that subsection VI, Reference 7 is deleted under this draft revision (see note 11).
11.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format guidance. Inclusion of cited SRP Sections as references and/or listing them in the references subsection is unnecessary since the SRP (NUREG-0800) is one document.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
12.	Editorial	Removed application content recommendations (apparently from Reg. Guide 1.70) to more clearly list specific areas of staff reviews.
13.	No change proposed.	Based upon PI 22999, the evolutionary plant design issues related to decoupling the OBE from the SSE and elimination of the OBE from design load combinations will primarily be addressed in revisions to other SRP Sections but may suggest SRP Section 3.9.5 editorial changes. The analyst evaluated the existing SRP Section 3.9.5 wording (esp. "as appropriate") and determined that it provides sufficient flexibility to remain consistent with implementation of any changes to other SRP Sections.
14.	SRP-UDP format item	Added Review Interface subsection of Areas of Review using numbered paragraphs to be consistent with SRP-UDP required format so that reviews performed by the SRP Section 3.9.5 PRB in other SRP Sections which are relevant to the overall review of reactor internals are detailed in their own subsection.
15.	**Integrated Impact 292**	Added Review Interface to SRP Section 3.6.2 since II 292 recommends modification of Review Procedures to clarify that asymmetric loadings should be considered. Pipe rupture locations producing design basis asymmetric loadings are reviewed by EMEB under SRP Section 3.6.2.
16.	PI 24332	Added review interface to SRP section 3.9.1. The ABWR FSER indicates that the staff's evaluation of criteria for the design reactor internals under normal, upset, emergency, and faulted loading conditions is discussed in FSER sections 3.9.1, 3.9.2.4, and 3.9.3.1.
17.	**Integrated Impact 291**	Added Review Interface description to address reviews related to the 60-year design life of evolutionary plants including the verification of adequate conservativism/margin in the design to account for added cyclic effects on the fatigue resistance of materials used in reactor internals (including bolting) over this longer life. SRP Section 3.9.1 is the section where the adequacy of identification of service lifetime cyclic loadings for certain Code class components and supports is reviewed. No review procedures are added to SRP section 3.9.5 as recommended in the II since criteria and details of Code class component and support fatigue design reviews are to be located in SRP sections 3.9.1 and 3.9.3.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
18.	PI 24332	Added review interface to SRP section 3.9.2. The ABWR FSER indicates that the staff's evaluation of criteria for the design reactor internals under normal, upset, emergency, and faulted loading conditions is discussed in FSER sections 3.9.1, 3.9.2.4, and 3.9.3.1. Also added this Review Interface to reflect existing SRP Section 3.9.5 references to reviews conducted under SRP section 3.9.2.
19.	PI 24332	Added review interface to SRP section 3.9.3. The ABWR FSER indicates that the staff's evaluation of criteria for the design reactor internals under normal, upset, emergency, and faulted loading conditions is discussed in FSER sections 3.9.1, 3.9.2.4, and 3.9.3.1. Also added this Review Interface to reflect existing SRP Section 3.9.5 references to reviews conducted under SRP section 3.9.3.
20.	**Integrated Impact 291**	Added Review Interface description to address reviews related to the 60-year design life of evolutionary plants including the verification of adequate conservativism/margin in the design to account for environmental and added cyclic effects on the fatigue resistance of materials used in reactor internals (including bolting) over this longer life. SRP Section 3.9.3 is the section where the fatigue properties of Code class components and supports are reviewed in detail. No review procedures are added to SRP section 3.9.5 as recommended in the II since criteria and details of fatigue design reviews are to be located in SRP section 3.9.3.
21.	SRP-UDP format item	Added Review Interface to SRP Section 3.9.4 consistent with existing SRP Section 3.9.5 references to that section.
22.	Current PRB names and abbreviations	Editorial change made to reflect the current SRP Section 3.9.5 PRB abbreviation for the Mechanical Engineering Branch.
23.	Current PRB names and abbreviations	Editorial changes made to reflect the current SRP Section 4.2 PRB name and abbreviation for the Reactor Systems Branch.
24.	Editorial	Editorial change made to improve grammar.
25.	Current PRB names and abbreviations	Editorial changes made to reflect the current SRP Section 4.5.2 PRB name and abbreviation.

Item	Source	Description
26.	**Integrated Impact 292**	Added Review Interface to new SRP Section 3.6.3 (see IPD 7.0 Form 3.6.2-1) since consideration of asymmetric loadings on reactor internals may not be required depending upon the adequacy of leak-before-break analyses justifying exclusion of such loadings from design bases.
27.	SRP-UDP Integration of Bolting Issues, Potential Impact 25749	Added a review interface reflecting reviews of bolting and threaded fastener programs under new SRP Section 3.13.
28.	Editorial	Revised wording to reflect applicability to all other SRP sections discussed in the review interfaces.
29.	Current PRB names and abbreviations	Editorial change made to reflect the current SRP Section 3.9.5 PRB abbreviation for the Mechanical Engineering Branch.
30.	Editorial	Grammar improvement-pluralized pronoun and verb since two regulations are discussed.
31.	Editorial	Grammar improvement-pluralized pronoun.
32.	**Integrated Impact 292**	Added discussion of asymmetric load considerations related to GDC 4. Also added discussion of modified GDC 4 provisions allowing leak before break analyses in lieu of design bases for certain postulated piping failures to support addition of Review Procedures verifying that asymmetric loadings on reactor internals are acceptably addressed in the design of reactor internals.
33.	Editorial	Revised to reflect the actual wording of GDC 10.
34.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format guidance. Inclusion of cited SRP Sections as references and/or listing them in the references subsection is unnecessary since the SRP (NUREG-0800) is one document.
35.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format guidance. Generally, only the first citation of a reference in an SRP Section is identified by reference number.
36.	Editorial.	Eliminated use of a gender specific pronoun.
37.	**Integrated Impact 292**	Added specific criteria and reference to NUREG-0609 to facilitate review of asymmetric loads on reactor internals where they are to be designed to accommodate such loads.

Item	Source	Description
38.	SRP-UDP format item.	Technical Rationale were developed and added for the following Acceptance Criteria: GCDs 1, 2, 4, and 10 and 10 CFR 50.55a. The SRP-UDP program requires that Technical Rationale be developed for the Acceptance Criteria.
39.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format guidance. Generally, only the first citation of a reference in an SRP Section is identified by reference number. Also clarified that subsection NG refers to the ASME Code.
40.	**Integrated Impact 292**	Modified Review Procedures to clarify that asymmetric loading conditions should be considered in the design of reactor internals if leak-before-break analyses justifying their exclusion from the design basis have not been approved.
41.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
42.	Editorial.	Eliminated use of a gender specific pronoun.
43.	Editorial	Grammar improvement.
44.	Editorial	Grammar improvement.
45.	Editorial	Grammar improvement.
46.	Editorial	Revised to more clearly reflect findings directly relating to the actual requirements of GDCs 2, 4, and 10 as they are applied to the review of reactor internals in subsection II of SRP Section 3.9.5.
47.	SRP-UDP Format Item	Editorial, standard change made to Evaluation Findings to address design certification reviews.
48.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
49.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
50.	SRP-UDP Guidance	Added standard paragraph to reflect existence of implementation information and schedules.
51.	SRP-UDP format item	Added or deleted references so that only those required are listed in this subsection per SRP-UDP format. Also rearranged the listing/numbering to place the references in the order specified per the SRP-UDP format.
52.	SRP-UDP format item	Added listing for 10 CFR 50.55a since it is cited in this SRP section as acceptance criteria.

Item	Source	Description
53.	Editorial	Revised to reflect the actual title of GDC 2.
54.	Editorial	Revised to reflect the actual title of GDC 4. Note that GDC 4 was amended, including changes to its title, subsequent to issuance of Revision 2 of SRP Section 3.9.5.
55.	**Integrated Impact 292**	Added identification of NUREG-0609 as a reference since changes based upon NUREG-0609 are proposed in SRP Section 3.9.5.
56.	SRP-UDP format item.	Format change to make the citation of references consistent with the SRP-UDP format guidance. Inclusion of cited SRP Sections as references and/or listing them in the references subsection is unnecessary since the SRP (NUREG-0800) is one document.

SRP Draft Section 3.9.5

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
290	Recommends revision of the SRP to address increased cumulative exposure of reactor internals to neutron irradiation over the 60 year design life of Evolutionary Plants.	SRP 3.9.5, Section III and IV. Note that since changes to the SRP based upon this position appear to represent type II changes outside the scope of the SRP-UDP, no changes to SRP 3.9.5 are incorporated in this draft revision.
291	Recommends revision of the SRP to address verification that design fatigue curves for Evolutionary Plant component and support materials (including reactor internals and bolting) provide adequate conservatism to address environmental and cumulative cyclic effects of reactor service conditions on the materials.	SRP 3.9.5, Section I, Review Interfaces EMEB-2 and 4 (with SRP sections 3.9.1 and 3.9.3).
292	Recommends revision of the SRP to address adequate design treatment of asymmetric blowdown loads on reactor internals due to postulated pipe ruptures for PWRs.	SRP 3.9.5, Section I, Review Interfaces EMEB-1 and other PRBs-2 (with SRP sections 3.6.2 and 3.6.3), Section II.3, Section II specific criteria item e, Section II Technical Rationale 3 second paragraph, Section III 4th paragraph, Section VI.7.
1247	Placeholder for possible future revision of the SRP (under the SRP maintenance program) to address a proposed rulemaking at 59 FR 52255 amending 10 CFR 100, 10 CFR 50.34, 10 CFR 50.54, and adding a new Appendix S to 10 CFR 50.	No changes in this draft revision.